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^{20}F POWER MEASUREMENT FOR GENERATION IV SODIUM FAST REACTORS

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ABSTRACT

The Phénix nuclear power plant has been a French Sodium Fast Reactor (SFR) prototype producing electrical power between 1973 and 2010. The power was monitored using ex-core neutron measurements. This kind of measurement instantly estimates the power but needs to be often calibrated with the heat balance thermodynamic measurement. Large safety and security margins have then been set not to derive above the nominal operating point. It is important for future SFR to reduce this margin and working closer to the nominal operating point. This work deals with the use of delayed gamma to measure the power. The main activation product contained in the primary sodium coolant is the ^{24}Na which is not convenient for neutron flux measurement due to its long decay period. The experimental study done at the Phénix reactor shows that the use of ^{20}F as power tagging agent gives a fast and accurate power measurement closed to the thermal balance measurement thanks to its high energy photon emission (1.634 MeV) and its short decay period (11 s).

1. Introduction

On SFR, power is monitored by ex-core neutron measurements using fission chambers located at the bottom of the primary vessel. This measurement instantly estimates the power but has to be periodically calibrated with the heat balance measurement. This calibration is used to set the nominal operating point regarding to the reactor safety and thermal hydraulics efficiency point of view. It should be noted that this measurement is only available after thermal hydraulics stability of primary and secondary circuits, and steam generators is achieved.

Drift between the measured ex-core neutron flux and the instant released thermal power (fission rate) is mainly due to modifications in sodium density, fissile nuclei concentration increasing in the fertile blanket and global fissile nuclei concentration decreasing during fuel burn-up. Thus, between each heat balance calibration, the reactor does not work close to nominal operating point, inducing safety and thermodynamic efficiency decrease. High dynamic range and high temperature fission chambers are under development to achieve more axial representativeness of the power measurement [1].

This paper presents a method based on activation product to measure the instant neutron power. This kind of method is already used in PWR using the ^{16}N as power tagging agent [2].

The first part of this work concerns the research of power tagging agents. Then, simulation study has been done to evaluate measurability using high resolution gamma spectrometry system. The third part deals with the experimental test done at the Phénix SFR prototype. And finally, thanks to these simulations and experimental studies, performances of such a system have been predicted.

2. Methods

2.1 Practicability

Primary sodium coolant contains corrosion products, fission products and activation products. In normal condition, only activation products resulting from the direct sodium coolant activation by fast neutron flux are experimentally measurable. The practicability of the power measurement depends on availability of an activation product which can be used as power tagging agent. Several characteristics are needed to be suitable for power measurement:

- The cross-section of the reaction producing the tagging agent has to be enough to achieve a significant concentration in the coolant.
- The second requirement is its radioactive constant. The decay period has to be short compared to primary cycle time in order to limit build-up effects. In the same time, its period should not to be too short compared to the sampling transit time to the measurement sample.
- Finally, high energy gamma emission is preferable to increase the signal-to-noise ratio in terms of gamma spectrometry measurement.

The main neutron activation reactions producing gamma emitters are shown in the table 1. Dissolved argon from the cover gas, is taken into account (concentration about 0.02 ppm) [3]. The main primary cycle time is about 2 minutes. Neither ^{22}Na nor ^{24}Na can be used due to their too long decay period compared to primary cycle time. In contrast $^{24\text{m}}\text{Na}$ with a decay period of 2.2 ms immediately disappears outside the reactor core. ^{20}F and ^{23}Ne are more convenient thanks to their decay period respectively of 11 and 23 s. So, potential power tagging agents exist for SFR power measurement.

The main requirement for a power monitoring system is the ability to measure a large range of power with a good accuracy and a fast response. These metrological considerations are only achieved in high count rate and high resolution configuration. The gamma spectrometry system used for delayed gamma power measurement needs to have high energy resolution and high count rate abilities. This is why a cryogenic Hyper Pure Germanium diode (HPGe) and an analyser with a high saturation point have to be used.

Nuclear reaction	Induced emitter	Decay period (s)	Photon energy (keV)	Emission rate
$^{23}\text{Na}(n,2n)$	^{22}Na	8.22×10^7	1274.5	1
$^{23}\text{Na}(n,\gamma)$	^{24}Na	5.39×10^4	1368.63	1
			2754.03	1
$^{23}\text{Na}(n,\gamma)$	$^{24\text{m}}\text{Na}$	2.02×10^{-2}	472.0	1
$^{23}\text{Na}(n,\alpha)$	^{20}F	1.10×10^1	1633.7	1
$^{23}\text{Na}(n,p)$	^{23}Ne	3.75×10^1	439.9	3.3×10^{-1}

Tab 1: Neutron activation reactions on the sodium coolant

2.2 The ADONIS system

Conventional gamma spectrometry systems are limited to perform this kind of measurement where the system needs to reach high input count rates while maintaining optimal energy resolution and metrological stability during activity changing. The ADONIS system allows this instrumental limitation to be broken.

ADONIS (Algorithmic Development framewOrk for Nuclear Instrumentation and Spectrometry) is a project under development at the CEA-LIST for high resolution and high count rates and on-line applications. It offers more stability and more flexibility in use than conventional methods for temporal gamma spectrometry measurement and analysis.



Fig 1. ADONIS electronics

Filtering methods are usually based on a triangular or trapezoidal impulse response linear filter where parameters are adjusted to pre-requirement pulses shape to optimize spectral resolutions and dynamic range [4]. Pile-up rejection and life-time correction units are also implemented to increase the high count rate ability. For time varying activity measurement, conventional systems use loss free counting methods where count rates are corrected in real time with instant evaluation of the system dead time [5]. These methods are complex to adjust and are intrinsically limited in instant life time accuracy associated with signal filtering with shaping linear filters. This is why a new filtering and smoothing method for HPGe diode signals has been developed in the ADONIS system [6]. The ADONIS infinite impulse response and nonlinear smoother uses intrinsic statistical properties of nuclear signals to always obtain an optimal energy resolution and count rate estimation whatever the experimental condition status and changes are. The ADONIS analyzer gives an event stream composed, for each pulse of accurate value of collected charges quantity, duration of collection of charges, and time separation with previous event. This triplet allows an accurate and robust estimation of the instant gamma activity. The system is then able to measure

count rates above 1 Mcps without pile-up rejection units using. It is also optimal for real time temporal analysis of the gamma spectrometry measurement [7]. Now, ADONIS is under development using the PING system [8] (see figure 1) and a new spectrum analysis method [9,10].

2.3 The experimental test

The first challenge of the experimental test was to carry out a gamma spectrometry of the Phénix sodium coolant with the ADONIS prototype. The second challenge was to measure the ^{20}F signal and test the power measurement method. This experiment was the test n°103 of the "ultimate testing" program of the Phénix reactor before its definitive close in January 2010 [11,12].

The Global Delayed Neutron Detection (DND/G) cell has been chosen as the best available location at the Phénix nuclear power plant. The DND/G system has been set in order to detect clad failure by delayed neutron emitters measurement [13]. This location is not optimal in our problematic but it has been chosen for our experiment for two main reasons:

- The sodium transit time from core outlet to the measurement sample is the shortest at the pool-type Phénix reactor (about 30 s), increasing the probability to measure the ^{20}F signal.
- Despite a core outlet to sampling area distance of 1.5 m, the 6 sampling points located near each intermediary heat exchanger allows to limit the sodium flow heterogeneity impact on measurements. Figure 2 shows the principle of the DND/G system.

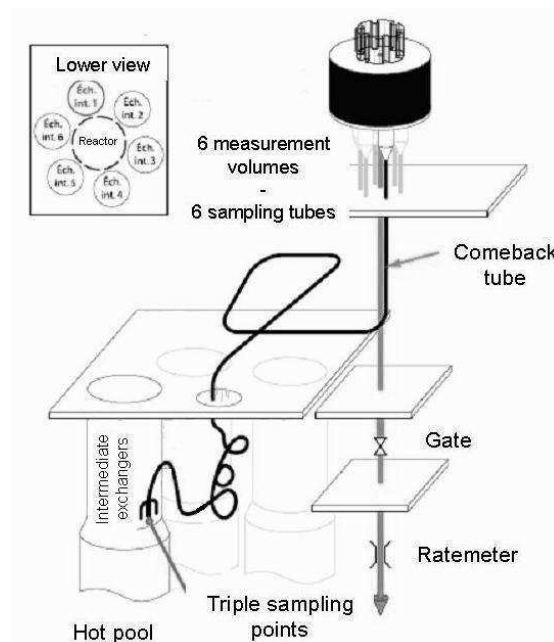


Fig 2. Principle schematics of the DND/G system

The gamma spectrometry system is composed of a reverse electrode coaxial germanium detector with a 10 % efficiency coupled with a reset transistor preamplifier and an hybrid cryostat (Cryo-Cycle) which is convenient for remote measurement [14,15]. HPGe diode signals have been processed and recorded using the ADONIS analyzer.

2.4 Simulation of the measurement

In order to optimize the measurement configuration in a pre-existing location, simulation studies have been useful to estimate signal and noise background magnitudes.

The following activation model has been used to obtain the order of magnitude of each activation product concentration at the reactor-core outlet. The concentration N_j of a activation product j in every coordinates in sodium core is calculated using the nuclear balance conservation equation as shown in equation 1.

$$\frac{\partial N_j}{\partial t} + \text{div}(N_j \vec{v}) = N_i \int_E \sigma_j(E) \phi(E) dE - \lambda_j N_j \quad (1)$$

Where N_i is the nuclear concentration of target nucleus i , $\sigma_j(E)$ is the reaction cross-section producing activation product j , λ_j is the decay constant of activation product j , \vec{v} is the velocity field of the sodium coolant, $\phi(E)$ is the neutron flux.

This equation is numerically resolved (axis discretization) using the mass flow rate conservation and the released heat calculation. The core geometry is radially meshed into three cylindrical parts (c dash) of equal neutron flux magnitude, sodium velocity and ^{23}Na atomic density namely: the fissile area, the fertile area and the reflector area. The axial distribution is modeled by a sinusoidal function $\phi_c(E, z) = \phi_c(E, \bar{h}) \sin\left(\frac{\pi z}{h}\right)$ where z is the

axial ordinate and h the fissile column high. This approximation is usually used as first-order model for axial flux distribution of PWR. This model could be far from reality in SFR case due to fertile blanket enrichment, but used in regards to the uncertainty associated to this pre-validation study. The mean activation product concentrations at the reactor core outlet ($z=h$) are finally weighted to obtain a global value (see equation 2).

$$N_{j,0} = \frac{\sum_c N_{j,c} S_c v_c}{\sum_c S_c v_c} \quad (2)$$

Where c is the area indicator (fissile, fertile, reflector), S_c is the equivalent sodium surface of each c area, $N_{j,0}$ is the global concentration of activation product j at the core outlet. During the sodium coolant cycle, radionuclides are diluted in primary coolant and could perform a complete coolant cycle if their decay period was long enough. This build-up effect is modelled by a series (equation 3) where n is the cycle number and τ_x the mean cycle time. The build-up transient stage and magnitude are important for ^{24}Na , ^{41}Ar and ^{22}Na . So, they can not be directly used to obtain a fast response power measurement. ^{23}Ne has a 5 % build-up magnitude with 4 minutes of transient state, ^{20}F has a neglected build-up impact (under 0.01 %). It could then be used to carry out, using its direct measurement, a fast and accurate power measurement.

$$\begin{aligned} N_{j,n} &= N_{j,0} \left(1 + \sum_{m=1}^n e^{-m\lambda_j \tau_x} \right) \\ \Rightarrow N_{j,\infty} &= N_{j,0} \left(1 + \frac{e^{-\lambda_j \tau_x}}{\lambda_j \tau_x} \right) \end{aligned} \quad (3)$$

A transfer function is then used to link the activation product concentration at the core outlet to the activity of each radionuclide contained in the measurement sample of the DND/G system. Experimental data extracted from the COLCHIX Phénix hot pool physical model (1:8 scale) have been used to simulate sodium flow [11]. Tagging agents are injected in each core assembly and analyzed using laser fluorimetry at each DND/G sampling point, which allows the thermal hydraulics impulse response F_j to be evaluated (see figure 3). Transit through the sampling tubes is considered as a plug flow with a constant transit time τ_m (see equation 4).

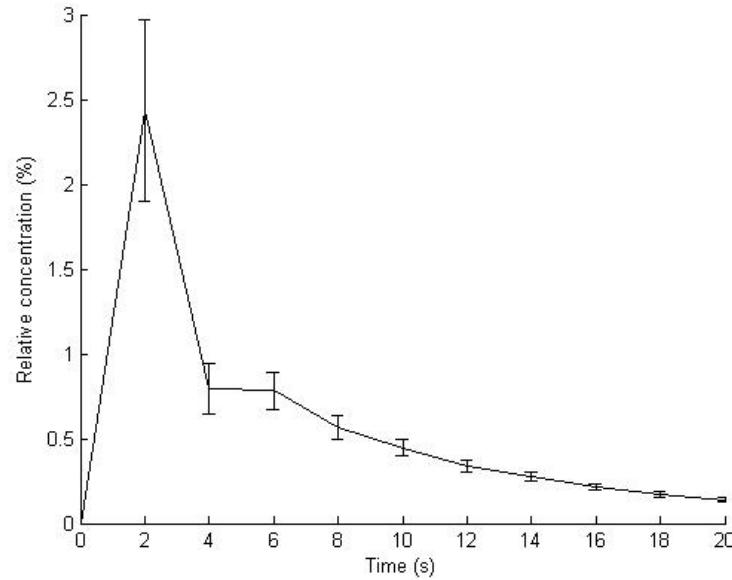


Fig 3. Sodium activation products impulse response

$$A_j = F_j N_j e^{-\lambda_j \tau_m} \quad (4)$$

The gamma spectrometry measurement is then simulated using MCNPX particles transport code validated in this field. The lead shielding thickness induces difficulties to carry out Monte-Carlo simulation. The simulation is then divided in two parts.

The first part of the simulation is a flux point tally (F5) to simulate the incident gamma flux yield on the HPGe detector. These simulations were strongly accelerated using variance reduction techniques to compensate statistical losses in the DND/G lead shielding. The exponential transform, stretching the path length between collisions, has been used to increase photon events outside the DND/G device. Moreover, geometry splitting and Russian roulette techniques were used to increase photon events in the detector direction.

The second part is the transfer of the obtained gamma flux in a surface source to managed a pulse height tally simulation (F8) of the HPGe sensor. Pulse height tally simulations have been experimentally validated showing efficiency overestimation due to recombinaison phenomena [12].

First simulations show that the ^{20}F signal is too low to be experimentally measured because of the lead shielding of the DND/G system. Operations have then to be done. The HPGe sensor is then shielded with a lead castle in order to increase the source signal to the diffused signal ratio and a collimation hole has been designed to increase the incident flux from the source in regards to not impact the security level of the DND/G system (safety system).

3. Results

3.1 Simulation results

The figure 4 is a simulated spectrum of the experiment at 350 MWth. The gamma signal is mainly composed of the ^{24}Na signal with photoelectric peaks at 2.754 MeV and 1.369 MeV and escape peaks at 2.243 MeV and 1.732 MeV. The probability to measure the ^{20}F signal has been increased by the operation but the statistical accuracy still far from the optimal situation. ^{22}Na , ^{41}Ar , ^{23}Ne is not measurable.

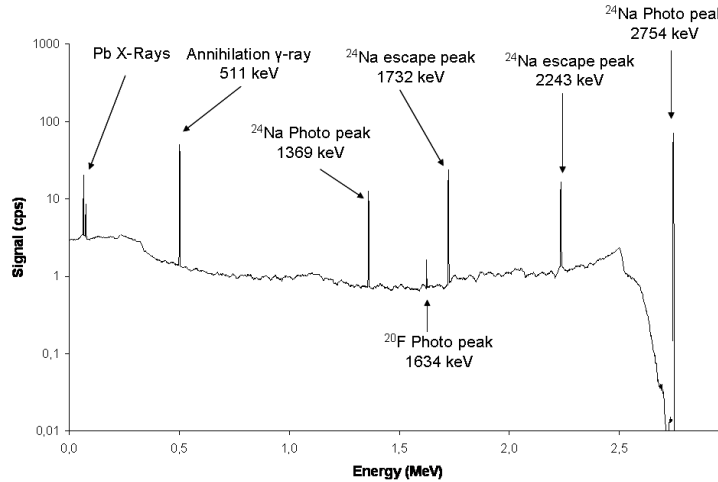


Fig 4. Simulated gamma spectrum of the experiment (350 MWth)

3.2 Experimental results

The system was installed in April 2009 for the ultimate testing program start-up in May 2009. Thirty hours of data has been recorded during reactor working operations.

The figure 5 shows a spectrum obtained at 302.5 MWth. As predicted by simulation, an important scattering signal is observed with lead X-rays peaks and an important annihilation peak due to lead shielding. The scattering signal is higher in the experimental case, especially at low energy, due to limitation of the particle tracking in respect with calculation time. The ^{20}F signal at 302.5 MWth is measured with a count rate of 0.84 ± 0.11 cps for a ^{24}Na and a Compton background of 1.75 cps.

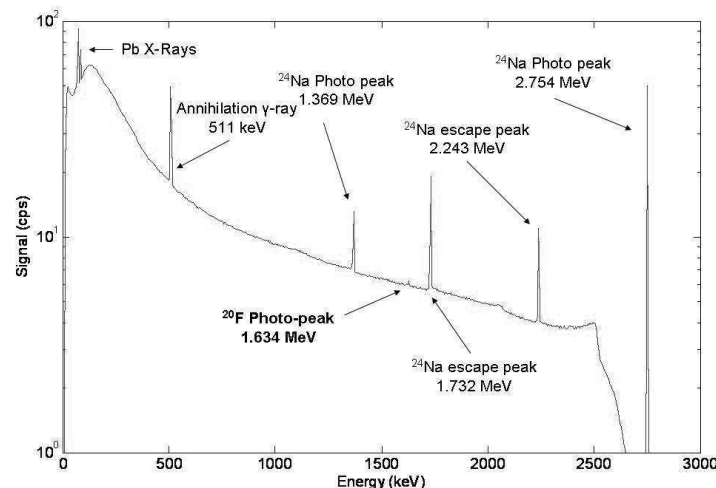


Fig 5. Measured gamma spectrum at 302.5 MWth

During the last part of the power increase (May 25th), three accurate thermal balance power measurements were carried out. The figure 6 shows the power measurement of the ex-core fission chambers (continue curve) and the power measurement calculated by thermal balance (triangular points). During this first power increase, no thermal balance calibration has been done. These reference measurements are used to study the ^{20}F signal behavior with power.

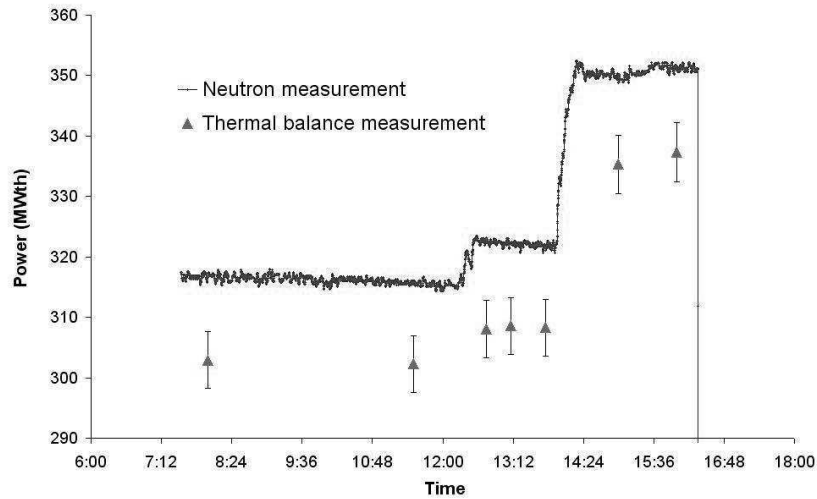


Fig 6. Power increase at May 25th

The ^{20}F and ^{24}Na measured signals are compared with the thermal reference power measurements. The temporal integration of the signal is equal to twenty minutes (as the thermal reference power measurement). Figures 7 and 8 respectively show ratio between thermal power versus ^{20}F signal and ratio between thermal powers versus ^{24}Na signal for each measurement point.

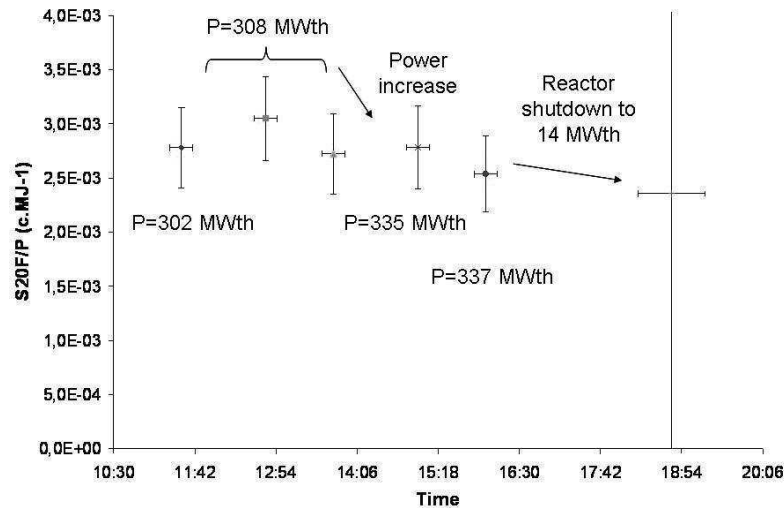


Fig 7. ^{20}F signal to released power ratio at the reference power measurement points

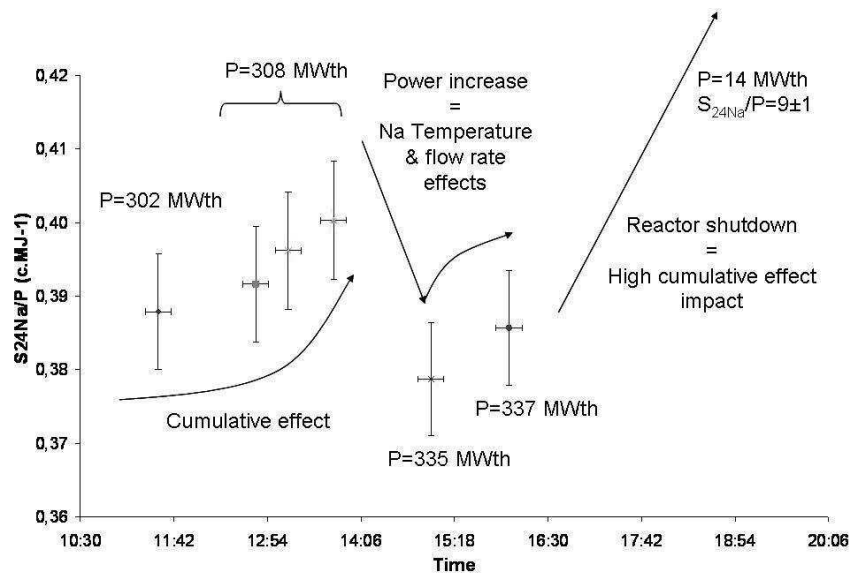


Fig 8. ^{24}Na signal to released power ratio at the reference power measurement points

The ^{20}F signal has been estimated at the reference points within 12 % of statistical error (one standard deviation). Nevertheless the ^{20}F signal shows a linear behavior with thermal power in this range of uncertainty. The ^{24}Na signal has been obtained with 2.5 % of statistical error allowing us to study effects of reactor status on the activation products activity. The figure 8 shows that:

- The ^{24}Na signal is strongly impacted by the build-up effect. An increasing of the signal during a constant power stage 12:00-14:00 and 15:00-16:30 is observed. After 16:30, the power suddenly decreases less than 50 MWth but a quasi-constant ^{24}Na activity measured.
- A significant decrease [3;10] % of the ^{24}Na signal is observed at 15:00. This falling of activity is correlated to the flow rate increase +7.0 %, the core temperature increase +3.3 % and the sampling temperature increase +4.0 %.

The figure 7 shows that no build-up effect has been measured in ^{20}F case. The poor counting statistics of the measurement does not allow us to study the reactor parameters impact on the ^{20}F response. Theoretically, the ^{20}F activity should be influenced, as the ^{24}Na , by the sodium flow rate and density changing with temperature.

These considerations show that the ^{20}F power measurement will be limited in accuracy during power increasing stage without real time corrections. At nominal power sodium flow rate and temperatures are quasi-constant. Burn-up and breeding phenomena are then the only remaining parameters that could impact the systems response.

3.3 Perspectives

The poor precision of the ^{20}F power measurement is due to the non-optimal configuration at the Phénix reactor. A ^{20}F power measurement system could be set-up (transit time and sample volume) in order to obtain a high statistical level for nominal power monitoring thanks to the high count rate and adaptive ADONIS system. Figure 9 shows activity evolution as a function of transit time to core outlet. In order to increase the ^{20}F signal and reduce the ^{24}Na Compton background, the transit time from reactor core to measurement sample has to be set as short as technically possible.

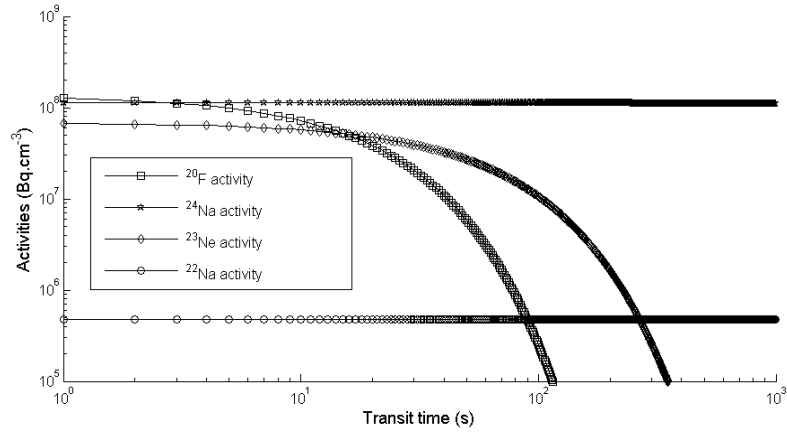


Fig 9. Activity evolution as a function of transit time in sodium sample

For instance: a ^{20}F power measurement system is simulated in pseudo-optimal conditions to get an estimation of the potential performance of such a system in terms of statistical accuracy and response time. The following characteristics are considered:

- a direct sampling at the core outlet,
- a transit time to measurement sample equal of 5 s,
- a mean cycle coolant time equal of 100 s,
- a coaxial HPGe sensor of 60 cm³ of active volume and 2.0 keV energy resolution at 1.634 MeV and 1.4 keV energy resolution at 440 keV at low count rate. It should be noted that the detector size could be also optimized.

Sodium sample are simulated setting the maximum of throughput at nominal power. Energy resolution degradations and pulse pile-up phenomena are taken into account (see figure 10).

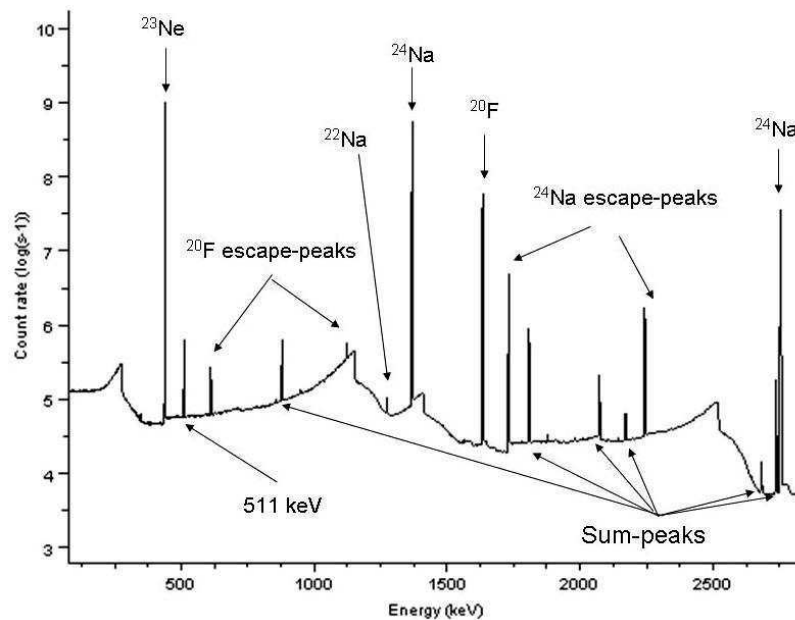


Fig 10. Simulated spectrum in optimal configuration on sodium sample

This simulation shows that 5 % of statistical error at nominal power could be achieved for less than 10 s of response time. Considering the only statistical point of view (^{24}Na Compton background), four neutron flux decades could be measured ($[1.10^{-4}; 1]$ NP).

4. Conclusion

This study shows that the ^{20}F produced by (n, α) capture on ^{23}Na nuclei could be used as power tagging agents. Its short decay period induces low build-up phenomenon and then allows a fast response system to be achieved. Experimental test at the Phénix reactor shows that the 1.634 MeV photons can be measured directly on sodium coolant sample by high resolution and high count rate abilities gamma spectrometry system.

This preliminary study also shows limitations of the ^{20}F power measurement:

- the build-up effect which induces a physical integration time,
- the Poisson statistics which induces a statistical integration time,
- the temperature and flow rate dependency,
- the burn-up and breeding dependency.

The build-up effect decreases as a function of the radioactive decay period of the tagging agent. Simulations and Phénix experiments show that the ^{20}F build-up impact could be very low. Other simulations (based on complete sodium coolant thermal hydraulics simulation) and other experimental tests have to be done to improve build-up understanding.

To reduce the Poisson statistical fluctuations, a high resolution and high count rates abilities system is needed. By setting the sampling transit time less than 5 s, an ADONIS system could achieved less than 5 % of statistical error at nominal power measuring ^{20}F with a short response time (less than 10 s). Considering the only the statistical point of view (^{24}Na Compton background), four neutron flux decades could be measured ($[1.10^{-4};1]$ NP).

However the experimental test at the Phénix reactor (^{24}Na analysis) shows an important sodium temperature and flow rate dependency during power increase stage. Theoretical studies show that burn-up and breeding phenomena could change (spectrum hardening or smoothing) the power response at nominal operating point (theoretically less than PWR case). A dual ^{20}F and ^{23}Ne measurement could be considered to correct these dependencies (flow rate impact during power increase and burn-up impact at nominal operating point). Other experimental tests and simulations (burn-up and breeding impact) should be done to evaluate the stability of a delayed gamma power measurement system for improve SFR safety and efficiency.

5. Acknowledgment

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